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17x17 Next Generation Fuel (17x17 NGF) Reference Core Report

Westinghouse

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1.0 Introduction and Summary

1.1 Introduction

This topical report presents generic information relative to a combination of improved fuel design features being introduced by Westinghouse and referred to herein as the 17x17 Next Generation Fuel (17x17 NGF) assembly design.

The features incorporated in this design include:

- 0.374 inch outer diameter fuel rods
- Axial blanket pellets (including annular blankets)
- Westinghouse Integral Top Nozzle (WIN) with removable top nozzle (RTN) feature
- High burnup Reduced Rod Bow (RRB) Top Grid
- Mid-grids with an I-spring rod support
- Five Intermediate Flow Mixing (IFM) grids
- New mixing vane design for mid-grids and IFM grids
- High burnup Bottom Grid
- Thick Wall Thimble and Instrumentation Tubes
- Tube-in-Tube Guide Thimble Assembly
- Debris Filter Bottom Nozzle (DFBN) with reduced pressure drop features
- Protective Grid (P-Grid)
- Oxide coating for mitigation of debris related failures
- Optimized ZIRLO™ fuel rod cladding

In addition, the fuel rod will potentially incorporate fuel rod plenum spring clips (licensed in WCAP-12610-P-A[(1)]), a stack height adjustment (potential removal of a pellet - ~ 0.75" reduction), and burnable absorber variations to meet specific rod internal pressure requirements based on burnup and power level conditions. These design aspects are not included in the overall discussion of this topical, but would be addressed in a plant specific analysis application.

This topical report provides a licensing basis for evaluating the 17x17 NGF fuel assembly design and, once approved, will serve as the basis for applications incorporating 17x17 NGF design features.

The 17x17 NGF fuel assembly is designed as a follow-on to the 17x17 Robust Fuel Assembly design (both RFA and RFA-2, hereafter referred to as RFA). Various 17x17 fuel plants were reviewed with respect to NSSS conditions to determine that the methodology used for the analysis is applicable for the 17x17 NGF fuel design and that the 17x17 NGF fuel design does not invalidate or change the analysis methodology (refer to Table 1-1 for NSSS conditions that have been reviewed). This does not preclude a plant, with NSSS conditions that differ somewhat from those listed in Table 1-1, from applying the 17x17 NGF design. Plant specific analyses/evaluations will be done for each initial application of 17x17 NGF. These analyses/evaluations will address the transition core effects from the existing 17x17 fuel product to a full core of 17x17 NGF.

Although the analyses presented do not in all cases constitute a bounding analysis, efficiencies in the review process for plant specific applications will be achieved by referencing much of the material contained in this generic application. Where plant specific analyses are required, these will be documented for that application with appropriate reference to the materials contained herein.

This topical report presents the 17x17 NGF design evaluation in conformance with the content guide given in the NRC Standard Review Plan (NUREG-0800)[2], refer to Table 1-2. As appropriate, reference is made to any materials already approved by the NRC. The report is organized along functional lines, consistent with the sub-chapters of a typical FSAR (i.e., Section 2.0 – Next Generation Fuel (NGF) Mechanical Design, Section 3.0 - Nuclear Design, Section 4.0 - Thermal and Hydraulic Design, Section 5.0 - Accident Analyses – Non-LOCA and LOCA, Section 6.0 - Reactor Vessel and Internals Evaluation, and Section 7.0 - Radiological Assessment) which support the 17x17 NGF design.

A brief summary of the 17x17 NGF design features follows. The features and figures, illustrating the design details, are presented in Section 2.0.

The 0.374 inch outer diameter fuel rods were used for the initial 17x17 STANDARD fuel design and were used again as part of the 17x17 VANTAGE 5H[3] (V5H) design. This design feature was transitioned into the current 17x17 RFA[4][5][6][7][8][9] design through the Fuel Criteria Evaluation Process (FCEP)[10] with design changes that were developed to rectify some known deficiencies with the 17x17 V5H design. The same fuel rod design will be used in the 17x17 NGF design except for the change to Optimized ZIRLO™ cladding[11]. There are no other changes, with respect to the fuel rod, from the previously licensed design.

The use of axial blanket pellets (including annular blankets) was introduced as part of the 17x17 V5H[3] design (VANTAGE + design for annular blankets[1]). Again, the same fuel rod design will be used in the 17x17 NGF design. Thus, there are no changes, with respect to the fuel rod, from the previously licensed design. Note: there was a change in the typical nominal length of the blankets from 6 inches to 8 inches. This change was developed to provide more margin in the fuel rod design to accommodate fission gas releases. This change was made under FCEP[12].

The Westinghouse Integral Top Nozzle (WIN) with removable top nozzle (RTN) feature will be used in the 17x17 NGF design (refer to Figure 2-2). It is essentially the same as the current WIN top nozzle that was developed for the 15x15 and 17x17 fuel designs. The WIN nozzle was developed to rectify issues associated with Primary Water Stress Corrosion Cracking (PWSCC) of the top nozzle holddown spring clamp screws. The modification to the 17x17 NGF WIN top nozzle is a slight change in the holddown springs to accommodate higher lift forces associated with the 17x17 NGF design relative to the existing 17x17 RFA fuel assembly. The original WIN top nozzle was introduced under FCEP[13].

The high burnup reduced rod bow (RRB) Inconel top grid was developed as part of the 17x17 VANTAGE 5/VANTAGE 5H[3] fuel design and has been incorporated on all Westinghouse fuel designs to date. The reduced rod bow design reduces the holding force between the inconel springs and the fuel rod at [

] a, c.

The low tin ZIRLO™ mid-grid design is a new feature with an "I-spring" rod support system (refer to Figure 2-7). This design was adopted from the CE 16x16 NGF grid design[14]. It is similar to the Westinghouse Optimized Fuel Assembly (OFA) spring design system. This new grid was specifically developed to improve the grid-to-rod fretting margin over current Westinghouse designs like 17x17 RFA[4][5][6][7][8][9]. The mixing vane design was modified from the 17x17 RFA mid-grid design and has a [ ] a, c to improve heat transfer performance during normal operations. Extensive CHF testing of the new mid-grid with its corresponding IFM-grid design was done at the Columbia University Heat Transfer Research Facility (HTRF). The CHF data and the corresponding correlation for the 17x17 NGF design has been submitted to the NRC for review and approval in WCAP-16766-P[15].

The 17x17 RFA fuel design used three IFM-grids. Each IFM-grid was located half way between the mid-grids in the upper three mid-grid spans. For the 17x17 NGF design, the top two mid-grid spans will have two IFM-grids equally spaced between the mid-grids (refer to Figure 2-1). The third mid-grid span will only have a single IFM equally spaced between the mid-grids. The five IFM-grid configuration is designed to provide additional turbulent mixing in these upper spans to add more thermal margin in the design to support plant uprates. The 17x17 NGF IFM grids will be fabricated from low tin ZIRLO™.

The high burnup Inconel bottom grid was developed as part of the 17x17 VANTAGE 5/VANTAGE 5H[3] fuel design and has been incorporated on all Westinghouse fuel designs to date. The high burnup Inconel bottom grid design increases the holding force between the inconel springs and the fuel rod to firmly hold the fuel rod throughout its life and ensure upward vertical growth.

The thicker-walled guide thimble tubes, along with the tube-in-tube guide thimble tube design, were developed to specifically address incomplete rod insertion issues (refer to Figure 2-3). These features were introduced with the modified V5H (MV5H) ZIRLO™ mid-grids in the 17x17 XL RFA fuel design. Both the thicker-walled guide thimble tubes and the tube-in-tube designs were dispositioned using FCEP[4][5][6][7][8][9] as these are features of the approved 17x17 XL RFA fuel design. Based on the in-reactor performance of the tube-in-tube design feature, this feature has also been implemented in the 15x15 Upgrade fuel design[16] and the revision to the 14x14 422 VANTAGE + fuel design[17]. The 17x17 NGF thicker-walled guide thimble tubes and the tube-in-tube designs will be fabricated from low tin ZIRLO™.

The original Debris Filter Bottom Nozzle (DFBN) was developed as part of the VANTAGE 5/ VANTAGE 5H[3] fuel assembly design. The design has not changed until now. The new design of the DFBN [ ] a, c of the assembly (refer to Figure 2-5). This partially offsets the [

] a, c.

The Inconel Protective Grid (P-grid) was developed in the early 1990s as a debris resistant feature designed to capture debris in the RCS and avoid debris induced fretting failures with the fuel. The P-grid is designed to [

] a, c. The design is in use in current Westinghouse fuel assembly designs. The P-grid was implemented under 10 CFR 50.59 prior to the approval of FCEP. The combination of the DFBN and P-grid are in compliance with the requirements of Generic Safety Issue (GSI) 191(18) and Generic Letter (GL) 2004-02(19). As documented in WCAP-16793(20), testing demonstrated with reasonable assurance that long-term core cooling will be achieved with debris in the recirculating coolant such that decay heat will be removed and a coolable core geometry will be maintained. The testing demonstrated that complete blockage of the fuel entrance with a large amount of fibrous and particulate debris is not observed with very conservative fibrous and particulate debris loadings. This position has been accepted by the NRC as noted in the Limitations and Conditions of the SER issued by the NRC for WCAP-16793(21).

The process of adding an oxide coating to the lower end of the fuel rod was developed in the early 1990s as another debris resistant feature that can be included in a fuel assembly design. Typically, the lower [ ] a, c inches of the fuel rods are given a thin oxide coating, similar to that obtained during the first few months of in-reactor operation. This oxide coating provides a harder surface for debris to fret against compared to the fresh cladding surface. The oxide coating process was implemented under 10 CFR 50.59, in conjunction with the P-grid and longer fuel rod bottom end-plug, prior to the approval of FCEP. These three features (i.e., DFBN, P-grid with longer end-plug and the oxide coated cladding) provide the maximum debris protection for the fuel rod from debris induced fretting.

Optimized ZIRLO™ cladding will be used in the 17x17 NGF design. Optimized ZIRLO™ cladding is a new feature that has recently been approved by the NRC(11). All SER requirements, specified by the NRC, will be accounted for in plant specific applications.

1.2 Summary

a. The results of the Mechanical Design evaluation performed for the 17x17 NGF fuel assembly design confirmed that:

- The 17x17 NGF fuel assembly design is mechanically compatible with the 17x17 RFA design, the reactor core components and internals, the movable incore detector system, and the fuel handling equipment.
- The design bases for fuel assembly structural components are satisfied.
- The design bases and limits for the 17x17 NGF fuel assembly and fuel rod performance are satisfied for lead rod average burnups of 62,000 MWD/MTU.
- The grid impact force for seismic and LOCA events were determined to be within the allowable limits as determined by grid crush tests.

- Hydraulic flow testing of the 17x17 NGF fuel assembly with the 17x17 RFA fuel design confirmed that fuel rod wear is within the Westinghouse established guidelines and provides additional margin relative to current designs like the 17x17 OFA or 17x17 RFA designs.

b. The results of the Nuclear Design evaluation performed for the 17x17 NGF fuel assembly design confirmed that:

- Standard nuclear design analytical models and methods accurately describe the neutronic behavior of the 17x17 NGF design.
- The 17x17 NGF nuclear design bases are satisfied.
- Safety limit characteristics of 17x17 RFA fuel design apply to the 17x17 NGF fuel design with no loss of margin to those limits.

c. The results of the Thermal and Hydraulic Design evaluation for the 17x17 NGF fuel assembly design confirmed that:

- With the implementation of the five IFM grids, thermal margins are increased. This margin can be used for improved fuel management, increased plant availability, uprates, and transition core effects.
- A transition core DNBR penalty on the 17x17 NGF fuel assembly, during the transition from 17x17 RFA, will be evaluated using the current evaluation methodology. The penalty is offset by the available DNB margin in reload designs.
- DNB testing of the 17x17 NGF fuel assembly led to the development of a new DNB correlation referred to as WNG-1 (WCAP-16766-P[15]). The 17x17 NGF correlation is not specifically covered in this topical report, but has been submitted to the NRC for review and approval.
- Hydraulic flow tests with the addition of the five IFM grids indicated a [ ] a, c increase in a 17x17 NGF core pressure drop compared to a 17x17 RFA core. The specific value is dependent on the features included in the 17x17 RFA fuel.
- No rod bow DNBR penalty is required in the mid-grid spans that include IFM-grids of the 17x17 NGF fuel assembly as the additional grids eliminate rod bow issues.

d. The results of the accident analysis/evaluation performed for the 17x17 NGF fuel assembly design confirmed that:

- For the non-LOCA accidents, the currently approved methods and computer codes used in the FSAR non-LOCA Chapter 15 analyses were found applicable for 17x17 NGF accident evaluations.
- For the LOCA analyses, the currently-approved realistic Large Break LOCA and deterministic Small Break LOCA methods are applicable for demonstrating that the acceptance criteria are met.

Table 1-1
NSSS Conditions Used for Scoping Analysis

a, c

Table 1-2
Standard Review Plan Section 4.2
Subsection II. - Acceptance Criteria

		SRP Subsection	Topical Report Section
Design Bases	**Fuel System Damage**	II.A.1.(a) - Stress, Strain or Loading Limits on grids, GT, fuel rods, control rods & other fuel system structural members	2.3.1.4, 2.4.1, 2.4.2, 2.4.4, 2.4.5, 2.5.2
		II.A.1.(b) - Strain Fatigue	2.5.7
		II.A.1.(c) - Fretting Wear	2.3.1.2, 2.5.6
		II.A.1.(d) - Oxidation, Hydriding and Crud	2.5.4
		II.A.1.(e) - Dimensional Growth, Rod Bow, Irradiation Growth	2.3.1.1, 2.5.9, 4.2
		II.A.1.(f) - Rod/BA Internal Gas Pressure	2.5.1, 2.5.10
		II.A.1.(g) - Holddown Forces	2.4.3, 4.1.2
		II.A.1.(h) - Control Rod Reactivity	6.3
	Fuel Rod Failure	II.A.2.(a) - Hydriding	2.5.4
		II.A.2.(b) - Cladding Collapse	2.5.8
		II.A.2.(c) - Fretting	2.3.1.2, 2.5.6
		II.A.2.(d) - Clad Overheating	2.5.5
		II.A.2.(e) - Pellet Overheating	2.5.5
		II.A.2.(f) - Excessive Fuel Enthalpy	5.1.3.d
		II.A.2.(g) - PCI	2.5.3
		II.A.2.(h) - Burst	2.5.9, 5.2
		II.A.2.(i) - Mechanical Fracturing	2.5.10
	Fuel Coolability	II.A.3.(a) - Cladding Embrittlement	2.5.10, 5.2
		II.A.3.(b) - Violent Expulsion of Fuel	5.1.3.d
		II.A.3.(c) - Clad Melting	2.5.5, 5.2
		II.A.3.(d) - Fuel Rod Ballooning	2.5.10, 5.2
		II.A.3.(e) - Structural Deformation (Seismic/LOCA)	2.3.1.3, 5.2.4.2, 6.2
Description & Design		II.B	1.1, 2.3
Design Evaluation		II.C.1 - Operating Experience	2.4.6
		II.C.2 - Prototype (LTA) Experience	2.4.6
		II.C.3 - Analytical Predictions	3.0 thru 7.0
Testing, Inspection and Surveillance Plans		II.D - Test, Inspections, Surveillance	Not required

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms. Fuel system damage includes fuel rod failure, which is discussed in subsection II.A.2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

The Testing, Inspection and Surveillance item has been marked as not required, since nothing in the analyses has identified a need for a new test, inspection or surveillance for the 17x17 NGF design. All standard test, inspections and surveillances specific in Technical Specifications and FSAR will still be done. Any tests, inspections or surveillances specified in other topical reports or SER will be completed as required.

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2.0 Next Generation Fuel (NGF) Mechanical Design

2.1 Introduction

The Standard Review Plan, specifically section SRP 4.2[2], provides the guidance for demonstrating the acceptability of a fuel design for use in-reactor. Table 1-2 provides an overview of those parameters that need to be addressed with a new fuel design. This section provides a discussion on many of the parameters discussed in SRP 4.2. Some of the parameters are discussed in the other analysis sections included in this topical report. The 17x17 NGF design has been designed [

] a, c. Not each criterion applies to each component, however, where applicable the SRP section will be called out and the appropriate criteria will be shown to be met.

2.2 Fuel System Design Description

The 17x17 NGF fuel assembly is designed to be mechanically compatible with the 17x17 RFA design for reactor operation with mixed fuel cores. The 17x17 NGF fuel assembly design data are given in Table 2-1. In this table, comparisons are made between 17x17 RFA and 17x17 NGF to show design similarities and differences.

Each 17x17 NGF fuel assembly consists of 264 fuel rods, 24 guide thimble tubes and 1 instrumentation tube arranged within a supporting skeleton structure. The instrumentation tube is located in the center position and provides a channel for insertion of an incore neutron detector when the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, or other core components, depending on the position of the particular fuel assembly in the core. Figure 2-1 shows a comparison of the 17x17 NGF fuel assembly relative to the 17x17 RFA design. The fuel rods are axially positioned within the fuel assembly structure to provide clearance between the fuel rod ends, and the top and bottom nozzles.

2.3 17x17 NGF Fuel Assembly

The design bases for the 17x17 NGF fuel assembly and each of the assembly components are identical to those for the VANTAGE + design as given in WCAP-12610-P-A[1]. The 17x17 NGF fuel assembly was designed (i.e., sized) to achieve up to a [ ] a, c MWD/MTU lead rod average burnup, consistent with the methodology in both WCAP-10125-P-A[22] and WCAP-15063-P-A[23], as supplemented to include Optimized ZIRLO™ cladding (WCAP-12610-P-A, Addendum 1[11]). Various features of the 17x17 NGF design, relative to the 17x17 RFA design, have licensing pedigree as noted below:

• 0.374 inch outer diameter fuel rods	WCAP-10444-P-A Addendum 2-A[3]
• Axial blanket pellets (including annular blankets)	WCAP-10444-P-A[3], WCAP-12610-P-A[1]
• Westinghouse Integral Top Nozzle (WIN) with removable top nozzle (RTN) feature	FCEP (LTR-NRC-04-23[13])

• High burnup Reduced Rod Bow (RRB) Top Grid	WCAP-10444-P-A(3)
• Mid-grids with an I-spring rod support	new feature (similar to the CE 16x16 NGF design(14))
• Five Intermediate Flow Mixing (IFM) grids	new feature (WCAP-10444-P-A(3) documented 3-IFMs)
• High burnup Bottom Grid	WCAP-10444-P-A(3)
• Thick Wall Guide Thimble and Instrumentation Tubes	FCEP (NSD-NRC-98-5796(7))
• Tube-in-Tube Guide Thimble Assembly	FCEP (NSD-NRC-98-5796(7), LTR-NRC-02-55(9))
• Debris Filter Bottom Nozzle (DFBN) with reduced pressure drop features	new feature (WCAP-10444-P-A(3) documented original DFBNs)
• Protective Grid (P-grid)	Implemented under 10 CFR 50.59 prior to FCEP
• Oxide coating for mitigation of debris related failures	Implemented under 10 CFR 50.59 prior to FCEP (Some plants already employ the oxide coated cladding feature. Thus, it may be new for some plants.)
• Optimized ZIRLO™ fuel rod cladding	WCAP-12610-P-A, Addendum 1-A(11)

2.3.1 Fuel Assembly Design Bases and Evaluations

2.3.1.1 Fuel Assembly Growth

Design Basis: Axial clearance between core plates and nozzle end plates should allow sufficient margin for fuel assembly and fuel rod irradiation growth to established design burnups.

Evaluation: This criterion assures that excessive forces on a fuel assembly will not be generated by the hard contact between the top nozzle and the upper core plate. Such forces could lead to fuel assembly bowing or guide thimble distortion. As such, the fuel assembly is typically sized to support growth up to a [ ] a, c MWD/MTU, even though the design will only be licensed to the current licensed burnup limit. Sizing accounts for the irradiation growth behavior of material used.

2.3.1.2 Fuel Assembly Hydraulic Stability

Design Basis: Flow through the assembly should not cause wear that exceeds the Westinghouse guideline of [ ] a, c.

Evaluation: The 17x17 NGF fuel assembly has been flow tested in the Westinghouse VIPER Loop adjacent to a 17x17 RFA fuel assembly. Results of these tests confirmed that the projected fuel rod wear due to contact with the mid-grids and IFM-grids is well within the Westinghouse guideline of limiting wear to less than [ ] a, c. Testing was also performed using the Westinghouse FACTS Loop to confirm the pressure drop characteristics across the entire assembly and individual components as well as verifying that the fuel assembly vibration amplitude was less than [

] [a, c]. Testing for resonant high frequency vibration of the mid-grid and IFM-grid straps was conducted using the Westinghouse VISTA Loop with no significant vibration response found.

2.3.1.3 Fuel Assembly Structural Integrity

Design Basis: The 17x17 NGF fuel assembly must maintain its structural integrity in response to seismic and LOCA loads.

Evaluation: Testing and analysis were performed on the fuel assembly components to verify that structural integrity would be maintained during seismic and LOCA loads. Dynamic crush testing of the 17x17 NGF mid-grids and IFM-grids found a comparable mid-grid structural performance and an improved IFM-grid structural performance relative to the 17x17 RFA mid-grids and IFM-grids. [

] [a, c].

2.3.1.4 Fuel Assembly Shipping and Handling Loads

Design Basis: The design acceleration limit for the 17x17 NGF fuel assembly handling and shipping loads is a minimum of 4g's.

Evaluation: Testing and analysis were performed on the fuel assembly to verify that shipping and handling load requirements were met. Section 2.4 gives more detail on what was done for the different components.

2.4 Structural Components Design Bases and Evaluations

2.4.1 Bottom Nozzle

The 17x17 NGF Debris Filter Bottom Nozzle (DFBN) is a slight variant on the current Westinghouse DFBN, which was designed for 4g shipping and handling loads. The 17x17 NGF DFBN has been optimized to minimize the pressure drop with [

] [a, c]. All other dimensions and features, including flow hole diameter, are identical to the current 17x17 DFBN design.

Design Basis: The bottom nozzle design bases are the same as those given in Section 2.3.1.2 of WCAP-10444-P-A[3].

Evaluation: Confirmatory testing was performed to verify the load-deflection characteristics and the flatness of the 17x17 NGF DFBN under 4g loading conditions. The test results met the design requirements.

2.4.2 Top Nozzle

The top nozzle assembly design for 17x17 NGF will be the Westinghouse Integral Nozzle (WIN) with the reconstitutable top nozzle (RTN) type (split insert with lock tube) joint connection for attachment to the skeleton assembly. The WIN design is being offered to all Westinghouse 15x15 and 17x17 plant customers. The 17x17 NGF WIN will share the same basic nozzle design as the current production 17x17 WIN, the key difference being in the hold-down spring design (described in Section 2.4.3).

The 17x17 NGF spring attachment is the integral spring pad type, whereby the spring leaves are captured within the nozzle and the clamp is integral (homogeneous) with the nozzle. Refer to Figure 2-2. This design eliminates the nickel alloy threaded fastener that can be susceptible to PWSCC. This new design has in-reactor experience which demonstrates its acceptability. The design provides additional margin to the spring to nozzle joint with regard to loose parts.

Design Basis: The top nozzle design bases are the same as those given in Section 2.3.2.2 of WCAP-10444-P-A[3].

Evaluation: Design changes to develop the 17x17 NGF version of the WIN do not impact previous analysis demonstrating that the criterion is met. Shipping and handling loads have been analyzed using FEA methods and the nozzle functionally tested. The top nozzle design was shown to meet all requirements.

2.4.3 Fuel Assembly Holddown Springs

All 17x17 NGF hold-down spring packs use leaves similar to the current WIN spring leaves used on the standard 15x15 and 17x17, 12-foot fuel assemblies. Due to increased flow forces, the 17x17 NGF assembly will require a 4-leaf spring pack. The 4-leaf spring pack is designed to produce greater holddown forces than the current WIN spring packs used on standard 12-foot fuel assemblies. Note that for lower flow plants, a 3-leaf spring pack will be used to prevent excessive fuel assembly holddown forces.

Design Basis: The design bases for the holddown springs are the same as those given in Section 2.3.3.2 of WCAP-10444-P-A[3].

Evaluation: Hydraulic tests were performed to obtain the necessary inputs to determine the required holddown force. Load deflection testing of the spring packs was completed to determine the actual spring load deflection characteristics. A final verification analysis using standard Westinghouse methodology will be performed for plant specific requirements to verify that holddown requirements are met.

2.4.4 Guide Thimbles and Instrumentation Tube

The 17x17 NGF guide thimble design will use the tube-in-tube guide thimble, which resembles the instrumentation tube with a constant outer and inner diameter over the entire length of the guide thimble tube. This design utilizes a separate dashpot tube with a slightly reduced diameter, which is inserted into the guide thimble assembly and bulged into place, refer to Figure 2-3. In the 17x17 NGF tube-in-tube guide thimble design, both the guide thimble and dashpot tubes are terminated with end-plugs welded prior to skeleton assembly. The 17x17 NGF tube-in-tube design concept is based on the 17x17 XL RFA tube-in-tube design(6). This design is being generally incorporated into all new Westinghouse designs since it adds significant margin to preclude incomplete rod insertion (IRI).

The instrumentation tube will be fabricated with the same OD and ID as the guide thimble tube. The length of the 17x17 NGF instrumentation tube is larger than the guide thimble tube length because the guide thimble ends below the top nozzle; whereas, the instrumentation tube has to engage the top nozzle above the counter bore chamfer.

Design Basis: The general guide thimble and instrumentation tube design bases are the same as those given in Section 2.3.4.2 of WCAP-10444-P-A(3).

Evaluation: The stresses on the 17x17 NGF guide thimble tube assemblies are bounded by the current 17x17 XL design. Stress analysis on the 17x17 NGF guide thimble tube show adequate margin on shipping and handling loads.

2.4.5 Joints and Connections

The 17x17 NGF design uses joints and connections that are similar to existing Westinghouse designs. The top nozzle to thimble joint uses a reconstitutable insert. The top-grid, bottom-grid, mid-grids, and IFM-grids are all bulged to the skeleton. The tube-in-tube design slightly impacts the bottom of the fuel assembly. A restraint bulge connects the dashpot tube to the thimble tube. The protective grid is joined using a spacer. The bottom nozzle is connected to each of the thimble tubes using a high strength thimble screw.

Design Basis: For events expected during the life of the fuel assembly, the resulting Condition I and II loads shall not cause permanent deformation at the joints or connections nor prevent the continued use of the fuel assembly for its design life. For accident and unanticipated events, the resulting Condition III and IV loads shall not cause any deformations that would prevent emergency cooling of the fuel or prevent the safe shutdown of the reactor. In addition, the loads resulting from shipping and handling shall not cause any deformations that would prevent the fuel assembly from meeting all the operating requirements for its design life.

Evaluations: Confirmatory testing was completed to verify the integrity of the joints and connections during the life of the fuel assembly, for any accident and unanticipated events and for any loads resulting from shipping and handling.

2.4.6 Grid Assemblies

Top and Bottom-Grids

The top and bottom grids, used on the 17x17 NGF design, are essentially identical to those used on the 17x17 RFA fuel design. The 17x17 NGF top and bottom grids have [ ] $^{a, c}$ straps. The grid provides 6-point rod support (2 vertical springs and 2 vertical dimple pairs per cell). Sleeves are brazed into the grid at thimble cell locations and these sleeves are then bulged to the thimble tube to fix the location of the grid. The 17x17 NGF top and bottom grids are unchanged from the 17x17 RFA grid design except for minor spring changes to ensure sufficient rod support over the life of the fuel. In addition, the Reduced Rod Bow (RRB) top grid [ ] $^{a, c}$.

Mid-Grid

The principal design difference between the 17x17 NGF mid-grid and the 17x17 RFA mid-grid is the implementation of the "I-spring" versus the diagonal spring. The 17x17 NGF mid-grid design has a 6-point rod support system (2 vertical springs and 2 horizontal dimple pairs per cell). Consistent with the 17x17 RFA design, 17x17 NGF uses [ ] $^{a, c}$ strap material is laser welded and has skeleton mid-grid attachment sleeves. These mid-grids have increased contact area and a modified 17x17 RFA mixing vane design. Refer to Figures 2-6 and 2-7.

Intermediate Flow Mixers (IFM)

The 17x17 NGF IFM design provides additional coolant turbulence in the high temperature spans near the top of the fuel assembly. Consistent with the 17x17 RFA design, 17x17 NGF uses [ ] $^{a, c}$ strap material is laser welded and has skeleton IFM grid attachment sleeves. Traditional IFM applications place one IFM near the mid-point of the upper three mid-grid to mid-grid spans. The 17x17 NGF design includes two IFMs in the upper two mid-grid to mid-grid spans and a single IFM in the third from the top mid-grid to mid-grid span, refer to Figure 2-1. They are spaced to create an equal length between grids that is not within the lateral projection of the grid's inner strap. These IFM grids have a modified 17x17 RFA mixing vane design. Refer to Figures 2-8 and 2-9.

Protective-Grid (P-grid)

The protective grid is unchanged from that which is currently used on 17x17 RFA fuel. The protective grid is a welded grid with [ ] $^{a, c}$ straps. The protective grid supports the fuel rod with four coplanar dimples in each cell. The primary function of the protective grid is debris mitigation. The protective grid accomplishes this by sectioning the flow holes in the bottom nozzle, stopping debris before it can reach the fuel rod cladding. Refer to Figure 2-10.

Design Basis: The grid design bases are the same as those given in Section 2.3.5.2 of WCAP-10444-P-A[(3)].

The grids must function acceptably under loading limits and not fail due to fatigue. In addition, the interaction between the grid and fuel rod should not result in conditions beyond the allowable fretting wear guidelines.

Evaluation: The 17x17 NGF mid-grids and IFM-grids have improved structural performance relative to the 17x17 RFA version of those grids. The evaluation of the 17x17 NGF grids is based on the extensive design and irradiation experience with previous grid designs and the component testing and analysis completed with the 17x17 NGF design.

Fatigue testing and analysis was satisfactorily completed for the rod support features.

The 17x17 NGF fuel assembly has been flow tested in the Westinghouse VIPER Loop adjacent to a 17x17 RFA fuel assembly. Results of these tests confirmed that the projected fuel rod wear due to contact with the mid-grids and IFM-grids is well within the Westinghouse guideline of limiting wear to less than [ ] a, c.

2.4.7 LTA Programs

17x17 NGF Lead Test Assemblies (LTAs) are currently in operation at two plants, [ ] a, c, as part of an irradiation demonstration program that will provide early confirmation performance data for the 17x17 NGF design including material performance of Optimized ZIRLO™ cladding. Eight 17x17 NGF LTAs were inserted at each plant. All 17x17 NGF LTAs have completed at least two cycles of irradiation demonstration. Three of the [ ] a, c twice burned LTAs were removed and inspected. The conclusion of the examination was that all of the parameter results, such as the maximum fuel rod and structural oxide thicknesses, fuel rod and fuel assembly growth, fuel rod creep, fuel assembly bow, and RCCA drag, etc, were within expected ranges. Four of the [ ] a, c twice burned LTAs were removed and inspected. Based on this inspection, it was also concluded that all the parameter results were within expected ranges. These examinations support the future use of the 17x17 NGF on a region and full core basis.

2.5 Fuel Rod Design Bases and Evaluations

The principal difference between the 17x17 NGF and the 17x17 RFA is in the fuel assembly features. The fuel rod design is not changed other than the cladding material, as noted in Section 1.1. Evaluations have been done to verify that the current licensed fuel rod bases and design criteria can be met for the 17x17 NGF design. The 17x17 NGF fuel rod design has been evaluated using the NRC-approved Westinghouse fuel rod performance code[23] to licensed burnup levels. The fuel rod design bases and criteria are described below. Those which are particularly affected by the above-mentioned 17x17 NGF considerations are so noted in the following sections.

The design bases and limits for the 17x17 NGF fuel, although somewhat redundant with respect to 17x17 RFA[4][5][6][7][8][9], are given in the following subsections. The design bases for 17x17 NGF fuel are identical to those for 17x17 RFA fuel. Refer to Figure 2-4.

The fuel rod will potentially incorporate variations to meet specific burnup and power level conditions (i.e., burnable absorber). These design aspects are not included in the overall discussion of this topical, however, for some plants, these features may be necessary to ensure that fuel rod design criteria would continue to be met. It will also be noted, that if these items are included in a specific plant's design, they would need to be properly addressed in the plant specific analyses.

2.5.1 Fuel Rod Internal Pressure and DNB-Propagation

Design Basis: The fuel system will not be damaged due to excessive fuel rod internal pressure.

During Condition I and II events, the internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward clad creep during steady state operation and (2) extensive DNB propagation to occur. This NRC-approved fuel rod internal pressure limit is justified in Reference 24.

Evaluation: The 17x17 NGF fuel rod internal pressures are evaluated in the same manner as for other Westinghouse fuel types. Gas inventories, gas temperature, and rod internal volumes are modeled and the resulting rod internal pressure is compared to the design limit. The design evaluations to licensed burnup levels verify that the fuel rod internal pressure as calculated, will meet the design basis.

Since the fuel rod design has not changed relative to the 17x17 RFA fuel rod design, both the current DNB-Propagation methodology[25] or the new DNB-Propagation methodology[26] will be applicable to the design.

2.5.2 Fuel Rod Clad Stress and Strain

Design Basis: The fuel system will not be damaged due to excessive fuel rod clad stress and strain.

The design limit for the fuel rod clad stress is that the volume average effective stress calculated with the von Mises equation considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling, uniform clad creep, and pressure differences is less than the 0.2% offset clad yield stress, with due consideration of temperature and irradiation effects under Condition I and II modes of operation. While the clad has some capability for accommodating plastic strain, the yield stress has been established as a conservative design limit.

The design limit for the fuel rod clad strain is the total plastic tensile creep strain due to uniform clad creep and uniform cylindrical fuel pellet expansion due to swelling and thermal expansion is less than 1% from the un-irradiated condition. The total tensile strain due to uniform cylindrical pellet thermal expansion during a transient is less than 1% from the pre-transient value. These limits are consistent with proven practice(22).

In the case of fuel rod cladding stress, an alternate approach would be to replace the classic clad stress criterion with a more recent licensed fuel rod clad stress criterion(27) consistent with NUREG-0800, SRP section 4.2(2) and based on the ASME pressure vessel guidelines.

Evaluation: The clad stresses in the 17x17 NGF fuel rod clad caused by power transients are evaluated using the same methods as are used for other Westinghouse fuel designs. Clad and pellet geometries, temperatures, and irradiation behavior are modeled, and the resulting clad stresses and clad strains are compared to the stress and strain limits. Design evaluations will show that the clad stresses and clad strains for the 17x17 NGF fuel rod designs (with and without burnable absorbers) can meet the design limits(22)(27).

2.5.3 Pellet Cladding Interaction

Design Basis: The fuel system will not be damaged due to excessive pellet cladding interaction (PCI).

The fuel rod cladding is protected against damage from PCI by limiting the cladding deformation due to pellet thermal expansion. This is done by limiting operation so that fuel centerline melt does not occur, and the cladding stress and strain are limited.

Evaluation: The fuel rods in the 17x17 NGF fuel assemblies are evaluated using the same methods as are used for other Westinghouse fuel designs. The specifics of fuel centerline melt are given in Section 2.5.5 and fuel rod cladding stress and strain in Section 2.5.2.

The Optimized ZIRLO™ cladding used in the 17x17 NGF fuel rods is expected to have the same or better PCI performance than standard ZIRLO™ cladding. The creep performance of Optimized ZIRLO™ has been shown to be the same as standard ZIRLO™ cladding. This results in the same rate of pellet-to-cladding gap closure as with the standard ZIRLO™ rod designs. The corrosion behavior has been shown to be better (lower oxide thickness), while the tensile strain capability is the same as standard ZIRLO™ cladding. This results in the same or better ability to absorb pellet induced strains with burnup as with standard ZIRLO™. The basis for these positions is given in Reference 11.

2.5.4 Fuel Clad Oxidation and Hydriding

Design Basis: Fuel rod damage will not occur due to excessive clad oxidation and hydriding.

In order to limit metal-oxide formation to acceptable values, the clad metal-oxide interface temperature is limited to licensed values for Condition I and Condition II transients, (ZIRLO™ and Optimized ZIRLO™ [ ] a, c (1)(11)). For Optimized ZIRLO™, the best estimate clad oxide thickness is limited to a licensed value of [ ] a, c (11). The clad hydrogen pickup is limited to a best estimate value of [ ] a, c (1)(11) at end of life to preclude loss of ductility due to hydrogen embrittlement by formation of zirconium hydride platelets.

Evaluation: The clad surface temperature, clad oxide thickness and hydriding of the 17x17 NGF fuel rod is evaluated by the same methods as are used for other Westinghouse fuel designs. The coolant temperature rise over the length of the fuel rod and temperature rise through the film, crud and oxide layer are calculated, and the temperature at the metal-oxide interface is determined[23]. The best estimate oxide thickness for ZIRLO™ cladding has been representatively shown to be less than [ ] a, c. Based on Reference 11, Optimized ZIRLO™ has been shown to have less oxidation than standard ZIRLO™. Therefore, the limit is met. The calculations show that the clad surface temperature and hydriding of the 17x17 NGF fuel rod meet the design limits.

2.5.5 Fuel and Clad Temperature

Design Basis: Fuel rod damage will not occur due to excessive fuel temperatures.

For Condition I and II events, the fuel system and protection system are designed to assure that a calculated centerline fuel temperature does not exceed the fuel melting temperature. The melting temperature of UO_2 is taken to be 5080 °F (un-irradiated) and to decrease by 58 °F per 10,000 MWD/MTU of fuel burnup.

Evaluation: The temperature of the 17x17 NGF fuel pellets is evaluated by the same methods as are used for all Westinghouse fuel designs. Rod geometries, thermal properties, heat fluxes, and temperature differences are modeled to calculate the temperature at the surface and centerline of the fuel pellets. Fuel centerline temperatures are calculated as a function of local power and rod burnup. To preclude fuel melting, the peak local power experienced during Condition I and II events can be limited to a maximum value which is sufficient to ensure that the fuel centerline temperatures remain below the melting temperature at all burnups. Design evaluations for Condition I and II events have shown that fuel melting will not occur for achievable local powers and licensed fuel rod burnup.

2.5.6 Fuel Clad Wear

Design Basis: The fuel system will not be damaged due to fuel rod clad fretting.

Westinghouse uses a design [ ]^{a,c} as a general guide in [ ]^{a,c} including fretting wear marks.

Evaluation: The predictive capability used to assess the wear characteristics of a 17x17 RFA assembly is also applicable to a 17x17 NGF assembly. The 17x17 NGF fuel assembly has been flow tested in the Westinghouse VIPER Loop adjacent to a 17x17 RFA fuel assembly. Results of these tests confirmed that the projected fuel rod wear due to contact with the mid-grids and IFM-grids is well within the Westinghouse guideline of limiting wear to less than [ ]^{a,c}. Thus, the design criteria with regard to clad fretting wear will be met for 17x17 NGF to its design burnup.

2.5.7 Fuel Clad Fatigue

Design Basis: The fuel system will not be damaged due to excessive clad fatigue.

The fatigue life usage factor is limited to less than 1.0 to prevent reaching the material fatigue limit.

Evaluation: Clad fatigue for the 17x17 NGF fuel rod design is evaluated by the same methods as are used for other Westinghouse fuel designs. Computer modeling of the fuel rod simulates a daily load follow cycling scheme 100-15-100% power and 12-3-6-3 hour intervals for residence times of more than 60 months. The potential impact of a wear scar on clad fatigue, consistent with limits imposed on the results of fuel assembly vibration test is to be done by incorporation of a stress concentration factor on the calculated stress amplitude. Design evaluations have shown that the cumulative fatigue usage factor, assuming a [ ]^{a,c} wear scar, will be met for the 17x17 NGF fuel.

2.5.8 Fuel Clad Flattening

Design Basis: Fuel rod failures will not occur due to clad flattening.

The fuel rod design should preclude clad flattening during projected exposure.

Evaluation: Calculations have been performed for 17x17 NGF fuel to show that predicted gap closure time does not exceed the required limit. The evaluation of the criteria is discussed in detail in Reference 28.

2.5.9 Fuel Rod Axial Growth

Design Basis: The fuel rods will be designed with adequate clearance between the fuel rod ends and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly.

Evaluation: The fuel assembly design will be sized to provide sufficient fuel rod-to-nozzle gaps. Sufficient top and bottom nozzle-to-fuel rod gaps are provided to assure that fuel rod growth is accommodated during the fuel design lifetime. The licensed methodology used to determine fuel rod growth for licensed fuel rod burnup is addressed in References 1 and 11.

The rod growth criterion has been verified using standard Westinghouse models and methods. The rod burnup capability with respect to the rod axial growth criterion is determined from the condition of no rod-to-nozzle interference, considering the worst case stack-up of the uncertainties in the rod-to-nozzle gap. This gap is obtained by calculating the maximum EOL rod growth, including growth and fluence model uncertainties, minimum EOL assembly growth and the minimum as-fabricated rod-to-nozzle gap, with the fabrication uncertainties calculated from the tolerances of all the relevant assembly components. Differential thermal expansion between the rod and the assembly is accounted for in the EOL rod-to-nozzle gap calculation. The design criterion is satisfied within a target peak rod burnup of [ ] a, c MWD/MTU with a design margin of [ ] a, c.

2.5.10 Fuel Rod Materials

Design Basis: The fuel rod design will use design values for properties of materials as given in References 1, 3, 10, 11 and 29, for ZIRLO™, Optimized ZIRLO™, IFBA and Gadolina material.

Evaluation: The material properties of the UO_2 fuel are not affected by the presence of a thin [ ] a, c ZrB_2 coating on the fuel pellet surface, therefore, the properties described in Reference 10 for UO_2 are also applicable, with due consideration to temperature and irradiation effects. In the ZrB_2 rod, the fuel pellets [

] $^{a, c}$ in B^{10}. The B^{10} acts as a burnable absorber. The irradiation behavior of the thin IFBA coating material has been reviewed and approved for use in Westinghouse PWRs in Reference 3.

Some material properties of the UO_2 fuel are slightly affected by the presence of Gadolinia in the fuel matrix, while other material properties are negligibly impacted. Reference 29 describes the appropriate material properties for Gd_2O_3 in UO_2, with due consideration to temperature and irradiation effects. In the Gadolinia fuel rod a small amount of Gd_2O_3 is mixed with the UO_2 and sintered together to act as a burnable absorber. The use of Gadolinia product has been reviewed and approved for use in Westinghouse PWRs in References 10 and 29.

ZIRLO™ is a modification of the Zircaloy-4 alloy. The comparative properties of the ZIRLO™ and Zircaloy-4 alloy are described in detail in Reference 1. Some of these properties, including density, thermal expansion, thermal conductivity and specific heat, have been verified in testing programs described therein. For all of the material properties other than specific heat, the properties of the two alloys are essentially identical, thus the same materials properties models are used for the ZIRLO™ alloy as are presently used for Zircaloy-4.

Optimized ZIRLO™ is a modification of the ZIRLO™ alloy. The comparative properties of the ZIRLO™ and Optimized ZIRLO™ alloy are described in detail in Reference 11. Some of these properties, including density, thermal expansion, thermal conductivity and specific heat, have been verified in testing programs described therein.

Table 2-1
17x17 RFA and 17x17 NGF Fuel Design Comparison

a, b, c

Table 2-1 (cont.)
17x17 RFA and 17x17 NGF Fuel Design Comparison

a, b, c

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Figure 2-1
Comparison of 17x17 NGF Design with 17x17 RFA Design

a, c

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Figure 2-2
17x17 NGF WIN Top Nozzle Design

a, c

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Figure 2-3
17x17 NGF Guide Thimble and Instrumentation Tube Connections

a, c

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Figure 2-4
17x17 NGF Fuel Rod

a, c

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Figure 2-5
17x17 NGF DFBN Flow Hole Design

a, c

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Figure 2-6
17x17 NGF Structural Mid-Grid

a, c

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Figure 2-7
17x17 NGF Mid-Grid "I-Spring" Design

a, c

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Figure 2-8
17x17 NGF Intermediate Flow Mixing (IFM) Grid

a, c

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Figure 2-9
17x17 NGF Intermediate Flow Mixing (IFM) Grid Design

a, c

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Figure 2-10
17x17 NGF Protective Grid/DFBN Configuration

a, c

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3.0 Nuclear Design

3.1 Design Bases

The design bases and functional requirements used in the nuclear design of the 17x17 Next Generation Fuel (NGF) cores are the same as those employed in previous Westinghouse 17x17 fuel designs such as the 17x17 VANTAGE 5H (V5H) design[3]. The VANTAGE 5H nuclear design bases are carried forward from the 17x17 VANTAGE 5 (V5) fuel design[3] and the 17x17 Optimized Fuel Assembly (OFA) design[24].

The 17x17 NGF design results in very small differences in nuclear design characteristics compared to prior 17x17 RFA fuel design. The primary nuclear design parameters such as fuel assembly pitch, fuel rod pitch, clad diameter, clad thickness, pellet diameter, and burnable absorber design are unchanged.

3.2 Design Methods

The methods and neutronics codes used in the 17x17 NGF core analysis require no changes to model the new fuel design. Applicable design methodology is described in Reference 30. Neutronics models are described in References 31, 32, and 33. These methods remain capable of modeling the 17x17 NGF design.

3.3 Design Evaluation

The neutronic characteristics of the 17x17 NGF design results are very similar to previous 17x17 fuel designs. The structural grid design is modified slightly from prior designs to use an I-spring rod support and a modified mixing vane geometry. Two additional Intermediate Flow Mixing (IFM) grids are added to the assembly slightly increasing the amount of structural material in the core region. The small increase in the amount of grid material in the core will be explicitly modeled in the core design models and has a very small effect on core reactivity and power distribution (i.e., much less than the differences seen in cycle-to-cycle fuel management variations).

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4.0 Thermal and Hydraulic Design

This section describes the thermal-hydraulic evaluation of the 17x17 Next Generation Fuel (NGF) design for general reload applications. The 17x17 NGF design improves heat transfer performance of the fuel design through the following design changes: (1) an enhanced mixing vane design used on both the mid-grids and Intermediate Flow Mixer (IFM) grids, and (2) a total of five IFM-grids used in the fuel assembly.

Similar to current Westinghouse fuel designs containing IFM-grids, the first IFM of the 17x17 NGF is placed in the middle of the grid span (i.e., between two mid-grids) at the mid-plane of the assembly. However, in the next two thermal-limiting grid spans downstream, the 17x17 NGF fuel assembly design contains two IFM-grids in each span, as compared to one IFM-grid for other fuel designs, e.g., 17x17 RFA.

The new design features of the 17x17 NGF for thermal improvement have been verified with respect to applicable thermal-hydraulic design criteria through testing and analysis. Included are discussions of the thermal-hydraulic design bases, effect of the design changes on rod bow evaluation, the design methods, and effect of mixed core on Departure from Nucleate Boiling Ratio (DNBR).

4.1 Thermal and Hydraulic Design Bases and Evaluation

The thermal and hydraulic design bases for the 17x17 NGF design are the same as those described in the Westinghouse Fuel Criteria Evaluation Process[10] described in this section. Each basis is followed by a discussion of the evaluation performed to verify that the basis is met.

4.1.1 DNB Design Basis

Design Basis: The fundamental criterion that must be met for core thermal-hydraulic design is the DNB design basis. SRP Sections 4.2[2] and 4.4[34] state that the DNB acceptance criterion provides assurance that there is at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB during normal operation or anticipated operational occurrence. Similar to all other Westinghouse fuel designs, the DNB design basis for the 17x17 NGF is that there will be at least a 95 percent probability at a 95 percent confidence level (95/95) that DNB will not occur on the limiting fuel rods during Condition I and II events.

Acceptance Limit: The DNB acceptance limit is the 95/95 DNBR limit defined by a DNB correlation applicable to the 17x17 NGF and approved by the NRC.

Evaluation: DNB tests (also referred to as Critical Heat Flux (CHF) tests) were performed with the new vane grids and the additional IFM-grids of the 17x17 NGF design at the Columbia University Heat Transfer Research Facility (HTRF) in 2003. In order to accurately reflect its thermal performance, a new DNB correlation was developed for the 17x17 NGF design based on the test results. The DNB correlation 95/95 DNBR limit and its applicable range are described in topical report WCAP-16766-P which has been submitted to the NRC for review and approval[15].

The DNB correlation will be used with the currently NRC-approved Westinghouse thermal design procedures[35][36][37] to ensure that the DNB design basis is satisfied in plant accident analyses for the 17x17 NGF fuel. The Westinghouse version of VIPRE-01 code[38][39], or other sub-channel codes qualified and approved with the DNB correlation, will be used for performing thermal-hydraulic analyses of the core and to calculate the minimum DNBR values in the hot channels. The plant analysis will account for uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters in addition to uncertainty in the DNB correlation.

4.1.2 Fuel Assembly Holddown Force

Design Basis: The fuel assembly will not be allowed to lift due to flow during normal operating conditions.

Acceptance Limit: The Westinghouse design limit is that the fuel assembly is designed to remain in contact with the lower core plate under all Condition I and II events with the exception of the turbine over-speed transient associated with a loss of external load[3][10][24].

Evaluation: The net upward force exerted on the fuel assembly is the result of the axial flow interacting with resistances along the flow path within a control volume. The upward hydraulic force of the 17x17 NGF design was calculated using the same method as for other Westinghouse fuel designs. The pressure loss coefficients used in the evaluation were determined from hydraulic tests of the 17x17 NGF design. The evaluation concludes that the 17x17 NGF design has sufficient holddown force margin to meet the acceptance limit. The fuel assembly holddown force margin will be verified for each plant application with plant specific core operating conditions.

4.1.3 Thermo-Hydrodynamic Stability

Design Basis: Operation under Condition I and II events will not lead to thermo-hydrodynamic instability in the reactor core. The types of instability considered are Ledinegg or flow excursion static instability and density wave dynamic instability.

Acceptance Limit: The Westinghouse design limits are that Ledinegg instability will not occur and that a large margin will exist to density wave instability[3][10][24].

Evaluation: For Westinghouse reactor designs, the Ledinegg instability is prevented because the slope of the reactor coolant system pressure drop versus flow rate curve is larger than the loop supply (pump head) pressure drop versus flow rate curve. The 17x17 NGF design will maintain the same characteristic of the reactor design.

The margin to the density wave instability will be evaluated using the method of Ishii[40] for the 17x17 NGF design, which is the same as for other Westinghouse fuel designs. An inception of this type of instability would typically require increases on the order of 100% or greater of rated reactor power. Margin to the instability will be verified for each plant application with plant specific core operating conditions.

4.2 Effect on Fuel Rod Bowing

The effect of 17x17 NGF rod bowing on DNB analysis has been evaluated using the same NRC-approved methodology[41] as for other fuel designs[1][3]. Rod bowing is dependent on span length between two grids and the rod moment of inertia. The five IFM-grids for 17x17 NGF decrease span length and provide additional fuel rod support. Based on the NRC-approved scaling method[41], the predicted channel closure in a grid span containing one or more IFM grids is less than 50 percent. The rod bow DNBR penalty is zero at less than 50 percent channel closure. The rod bow DNBR penalty in the non-IFM grid span will be offset by the same amount of DNBR margin retained in the plant specific DNB analysis.

4.3 Thermal and Hydraulic Design Methods

Besides a new DNB correlation for more accurate predictions of thermal margin, no other change in the thermal-hydraulic design methods currently used for other fuel designs is necessary for the incorporation of the 17x17 NGF design.

4.4 Transition Core DNBR Effect

There will be a DNBR penalty on 17x17 NGF in a mixed core with other 17x17 fuel products mainly due to the additional IFM grids of the 17x17 NGF fuel, as compared to the DNB analysis for a full core of 17x17 NGF. Since it is dependent on the amount of the 17x17 NGF fuel assemblies loaded in the core, the transition core DNBR penalty is determined on a plant specific basis. Using the NRC-approved evaluation methods[42], the transition core DNBR penalty for eight 17x17 NGF fuel assemblies in a mixed core with 185 - 17x17 RFA fuel assemblies in a 4-loop core is estimated to be less than [ ] a,c on the 17x17 NGF design. There is no penalty on the 17x17 RFA design in the mixed core with the 17x17 NGF. The penalty is smaller than the previous transition core DNBR penalty in a mixed core of a fuel design without IFM-grids and another fuel design containing IFM-grids. The transition core penalty will also be smaller for a mixed core containing more than eight 17x17 NGF fuel assemblies.

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5.0 Accident Analysis

5.1 Non-LOCA Safety Evaluation

5.1.1 Introduction and Overview

This section addresses the effect of the 17x17 NGF design on the non-LOCA accident analyses. This evaluation addresses the following NGF features:

- Westinghouse Integral Top Nozzle (WIN) with Removable Top Nozzle (RTN) feature,
- Mid-grids with an I-spring rod support,
- Five Intermediate Flow Mixing (IFM) grids (addition of two IFM grids to the assembly),
- New mixing vane design for mid-grids and IFM grids (modified RFA),
- Debris filter Bottom Nozzle (DFBN) with reduced pressure drop features,
- Protective grid,
- Oxide coating for mitigation of debris related failures (already used on some plants), and
- Optimized ZIRLO™ fuel rod cladding.

The revised mid-grid design and the addition of two IFM-grids are expected to improve the DNB performance in the thermally limiting section of the fuel. This is beneficial for the non-LOCA analyses. This will be taken into account in the implementation of the fuel in a plant application. Use of the assembly in a particular plant application increases the core pressure drop, possibly resulting in increased bypass flow. While this will not have a significant effect on the non-LOCA transients, this will also be addressed in the implementation.

Other features of 17x17 NGF are not addressed here but would have a negligible effect on non-LOCA analysis results as long as they do not adversely affect the DNB performance of the fuel. An evaluation of the effect of the use of Optimized ZIRLO™ cladding has been addressed in Reference 11.

Note that this assessment summarizes the expected impacts on the non-LOCA analyses. When 17x17 NGF is implemented at a given plant, the normal reload process[30] will be followed to address the impact (if any) of the fuel changes. The reload process is as described below:

- For every reload cycle, Westinghouse reviews the important fuel-related analysis assumptions to ensure that they remain bounding using the reload process. These assumptions include reactivity parameters, peaking factors, and fuel temperatures.

- The proposed fuel changes will increase the core pressure drop and could potentially affect the core bypass flow. While this will not yield significantly worse results in the non-LOCA analyses, it would be addressed in a plant specific analysis of the 17x17 NGF design.

- The plant Overtemperature and Overpower ΔT reactor trip setpoints (i.e., OTΔT and OPΔT) will be reviewed to ensure that they continue to protect against exceeding the core thermal limits.

- Potential transition core effects associated with the new fuel will be taken into account in plant and cycle-specific applications.

An assessment of the impact of the 17x17 NGF fuel design features on the various non-LOCA events is provided below, based on similar evaluations previously performed by Westinghouse.

5.1.2 Evaluation of Effects on Non-LOCA Computer Codes and Methods

The computer codes used in non-LOCA safety analysis consist of the LOFTRAN[43] or RETRAN[44] codes for calculating the NSSS transient response to accident events, the FACTRAN[45] or VIPRE[39] codes for hot rod fuel and clad temperature or heat flux evaluations, and the THINC[46] or VIPRE codes for the hot channel DNBR evaluation. In addition, the TWINKLE[47] code is used to calculate the core response for fast reactor transients where the RCS loop response is not important.

The system transient codes LOFTRAN and RETRAN use a detailed nodalization of the RCS primary side components (RCS hot and cold loops, reactor vessel, steam generator, pressurizer and reactor coolant pumps). In addition, they contain models of the reactor control and protection system, and engineered safeguards features. The core geometry is represented by several stacked axial nodes, with one set of stacked nodes for each RCS loop to facilitate inlet mixing. A simplified fuel rod radial heat transfer model is used in each node, which is calibrated to match a conservative set of fuel rod temperatures versus power. The core transient behavior is calculated with a point reactor kinetics model using pre-calculated kinetics coefficients (i.e., MTC, Doppler feedback, delayed neutron fraction, etc.). The core dynamic behavior is not sensitive to details of the fuel assembly design, and would be only very slightly affected by changes in the core pressure drop, flow rate, or core bypass caused by the implementation of the 17x17 NGF fuel assembly design.

The FACTRAN code uses a radial fuel pellet heat transfer model for calculating the transient temperature distribution in a cross-section of a fuel rod for a single axial node in the fuel channel. FACTRAN does not contain a detailed coolant thermal-hydraulics model. The FACTRAN code is used to calculate the hot channel average heat flux versus time for an external DNBR evaluation model such as THINC or VIPRE, or for calculating the hot spot fuel and clad temperature versus time with or without assuming DNB. FACTRAN includes the ability to input fuel or clad property models to take into account changes in materials properties. The FACTRAN calculation is not sensitive to the details of the fuel assembly design changes addressed here, and the results would only be slightly affected by the small changes in the core pressure drop, flow rate, or core bypass expected with the implementation of the 17x17 NGF fuel design.

The VIPRE code includes both a radial fuel pellet heat transfer model and a detailed multi-dimensional core thermal-hydraulics model. The VIPRE code may be used in place of FACTRAN to calculate the hot spot fuel and clad temperature versus time for certain transients with or without DNB. Changes in fuel or clad properties models can be taken into account using the code input. In addition, the THINC or VIPRE codes are used with a sub-channel model to perform a DNBR analysis for selected transients. The effect of the changes in the fuel assembly design addressed here are either insignificant or are taken into account as described in Section 4.4 of this report.

The TWINKLE code uses a finite-difference solution of the transient neutron diffusion equations with a relatively simple transient fuel and thermal-hydraulics model. It is used to calculate the core response for rapid reactivity insertion events (i.e., Rod Bank Withdrawal from Subcritical and Rod Ejection) where the RCS loop response is not important. The code is used in a one-dimensional model with multiple axial nodes representing the average core. The TWINKLE code models are not affected by the details of the fuel assembly design or the design changes which are addressed here.

In addition to the above, the RETRAN(44) and VIPRE(39) codes can also be used in combination with a kinetics version of the three-dimensional nuclear design code ANC(31), in a recently developed and approved methodology(48) for the more detailed analysis of non-LOCA accident events.

In summary, the computer codes and methods used in the non-LOCA safety analysis are essentially unaffected by the fuel assembly design changes addressed here, and remain valid for use in the evaluation of a plant implementing the 17x17 NGF fuel design.

5.1.3 Non-LOCA Accident Evaluation

This section provides a qualitative assessment of the expected effect of the 17x17 NGF fuel design changes on the non-LOCA analyses. The assessment will rely on previous experience with similar changes for Westinghouse plants. The discussion that follows is divided into sections based on the following classifications of non-LOCA events:

- Increase in Heat Removal by the Secondary System,
- Decrease in Heat Removal by the Secondary System,
- Decrease in Reactor Coolant Flow Rate,
- Reactivity and Power Distribution Anomalies, and
- Events Resulting in Increasing/Decreasing RCS Inventory.

5.1.3.1 Increase in Heat Removal by the Secondary System

A malfunction which causes an increase in heat removal by the secondary system results in a decrease in the temperature of the primary coolant. In the presence of a negative Moderator Temperature Coefficient (MTC), this can result in an increase in the core power level and a reduction in the minimum DNBR. In addition, if the malfunction is due to an increase in feedwater flow, this can cause overfilling of the steam generator.

The events typically analyzed for Westinghouse plants are:

- Feedwater System Malfunctions,
- Excessive Increase in Secondary Steam Flow, and
- Steamline Depressurization/Steamline Break events.

These transients are primarily "system-driven" in that the system transient results are not dictated by specifics of the fuel assembly geometry, but rather by the response of the RCS to the transient conditions. The details of the fuel assembly and fuel rod design are not modeled in the system transient and are not critical parameters.

The analyses of these events are performed to confirm that the primary coolant temperature reduction and associated insertion of positive reactivity does not result in an excessively large power increase that challenges the DNB limit for the plant. Although the DNB analysis of the fuel will be affected by this fuel change, the overall RCS statepoints (i.e., power, temperature, flow, pressure) will not be significantly different.

An evaluation will be performed to address the increase in vessel pressure drop and potential changes in core bypass flow and core stored energy. However, these changes will not have a significant effect on the results of the non-LOCA cool-down events.

With respect to DNB, the new mid-grid and IFM designs are expected to improve the DNB performance in the thermally limiting section of the fuel. An evaluation or analysis will be performed to quantify the effect of changes in the fuel assembly DNB performance on the results for the Feedwater Malfunction and Steamline Break DNB analyses. These changes will be seen in the Core Thermal Limits and the DNB correlation (e.g., WNG-1[(15)]) used in the analysis.

5.1.3.2 Decrease in Heat Removal by the Secondary System

A malfunction which causes a decrease in heat removal by the secondary system results in an increase in the temperature of the primary coolant. The heatup and expansion of the coolant can lead to a reduction in the DNBR, a primary or secondary system pressure increase, or pressurizer overfill.

The events typically analyzed for Westinghouse plants are:

- Loss of Electrical Load/Turbine Trip,
- Loss of Non-Emergency AC Power,
- Loss of Normal Feedwater,
- Feedwater System Pipe Break

As with the cool-down events, these events are primarily system-driven. The details of the fuel assembly and fuel rod are not modeled in the system transient and are not critical parameters.

For example, the Loss of Normal Feedwater/Feedwater Pipe Break events are driven by the heat transfer between the primary and secondary side and, in particular, the performance of the auxiliary feedwater system. The details of the fuel assembly and fuel rod are not modeled and are not critical parameters.

The analyses of these events are performed to confirm that limits on RCS pressure, pressurizer water volume, and secondary side pressure are met. Based on this, the fuel changes will not have a significant effect on the results of these analyses.

For a plant-specific application, an evaluation will be performed to address the consequences of an increase in vessel pressure drop and potential changes in core bypass flow and core stored energy. However, these changes will not have a significant effect on the results of the non-LOCA heatup events.

With respect to DNB, the new mid-grid and IFM designs are expected to improve the DNB performance in the thermally limiting section of the fuel. These changes will be seen in the Core Thermal Limits and the DNB correlation (e.g., WNG-1[15]) used in the analysis. An evaluation will be performed to confirm that the DNBR results for the Loss of Load event remain valid.

The Loss of Non-Emergency AC Power event can also result in a flow coastdown due a loss of power to the reactor coolant pumps. This is addressed in the section below.

5.1.3.3 Decrease in Reactor Coolant Flow Rate

A malfunction which causes a decrease in reactor coolant flow rate results in an increase in the temperature of the primary coolant in the core, and a decrease in the ability of the coolant to remove heat from the fuel. This can cause a reduction in the minimum DNBR and, in the case of a Locked Rotor event, a rapid increase in RCS pressure.

The events typically analyzed for Westinghouse plants are:

- Partial/Complete Loss of Forced Reactor Coolant Flow,
- Reactor Coolant Pump (RCP) Shaft Seizure or Shaft Break.

As with the other event categories addressed above, the RCS response to a loss of flow is "system-driven" in that the system transient results are not dictated by specifics of the fuel assembly geometry but rather by the response of the RCS to the transient conditions. The details of the fuel assembly and fuel rod design are not modeled in the system transient and are not critical parameters.

For a plant-specific application, an evaluation will be performed to address the consequences of an increase in vessel pressure drop and potential changes in core bypass flow and core stored energy. However, these changes will not have a significant effect on the system transient results of the decrease in RCS flow events.

With respect to DNB, the new mid-grid and IFM designs are expected to improve the DNB performance in the thermally limiting section of the fuel. These changes will be seen in the Core Thermal Limits and the DNB correlation (e.g., WNG-1[15]) used in the analysis. An evaluation will be performed to confirm that the DNBR results for the Loss of Flow and Locked Rotor events remain valid.

The peak cladding temperature (PCT) is calculated by Westinghouse for the Locked Rotor event. The FACTRAN or VIPRE computer code is used to calculate the PCT. As discussed above, the fuel design changes incorporated in the 17x17 NGF design are not expected to affect the PCT results. An evaluation will be performed to confirm that the PCT results remain valid in a plant-specific application.

5.1.3.4 Reactivity and Power Distribution Anomalies

Several non-LOCA transients are characterized by changes, either locally or globally, in core reactivity or power shape. The resulting increase in core power, or the core power peaking factor, could cause a reduction in the minimum DNBR. In the case of the RCCA Ejection event, the concern is the post-DNB pellet temperature and enthalpy increase.

The events typically analyzed for Westinghouse plants are:

- Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal,
- Dropped/Misaligned RCCA events,
- Uncontrolled Boron Dilution, and
- Spectrum of RCCA Ejection events.

The Rod Withdrawal at Power, Uncontrolled RCCA Withdrawal from a Subcritical Condition, and Dropped/Misaligned RCCA events are not expected to be significantly affected by the proposed fuel changes.

With respect to DNB, the new mid-grid and IFM designs are expected to improve the DNB performance in the thermally limiting section of the fuel. These changes will be seen in the Core Thermal Limits and the DNB correlation (e.g., WNG-1[(15)]) used in the analyses. An evaluation will be performed to confirm that the DNBR results for these events remain valid.

Changes in the overall RCS hydraulic parameters, such as core bypass flow and pressure drop, will also have to be evaluated but will not have a significant effect on the results of these analyses.

The Control Rod Ejection event is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The new mid-grid and IFM designs will not affect the analysis of this event since the primary concern is not DNB but the fuel pellet temperature and enthalpy.

Changes in the overall RCS hydraulic parameters will not significantly affect this analysis since the transient is over very quickly.

The Uncontrolled Boron Dilution event is the addition of unborated water to the RCS resulting in a positive reactivity insertion and erosion of plant shutdown margin.

The proposed fuel changes will not affect this analysis since the details of the fuel are not modeled.

However, the cycle specific RCS initial boron concentration, critical boron concentration, and shutdown requirements must be reviewed against the analysis assumptions to ensure that the results remain valid. This will be performed as part of the normal reload process.

5.1.3.5 Events Resulting in Increasing/Decreasing RCS Inventory

These non-LOCA events are characterized by either an increase or decrease in RCS water inventory. The events typically analyzed for Westinghouse plants are:

- RCS Depressurization, and
- Inadvertent Operation of the ECCS.

These transients are "system-driven" events and are not typically DNB limiting. Fuel details such as the cladding material, pellet density, and burnable absorber are not modeled in these analyses.

Therefore, the proposed fuel changes will not have a significant effect on the results of these analyses.

5.1.4 Conclusions

Based on the assessments provided above, the proposed fuel changes associated with 17x17 NGF will not have a significant effect on the non-LOCA analyses since it is expected that the DNB performance in the thermally limiting section of the fuel will improve due to the new mid-grid design and the addition of two extra IFM-grids. Some evaluations will be performed to address changes in the DNB performance and RCS hydraulic parameters. These will be addressed in a plant and cycle-specific application. These evaluations will demonstrate that implementation of the new fuel design does not result in any violations of the non-LOCA analysis acceptance criteria.

In addition to the event-specific evaluations described above, the normal reload process will be followed to ensure that the fuel-related analysis assumptions remain bounding. Additionally, the plant reactor trip setpoints (i.e., Overtemperature Delta-T (OTΔT) and Overpower Delta-T (OPΔT)) will be reviewed to ensure that they continue to protect against exceeding the core limits.

5.2 LOCA

5.2.1 Introduction and Overview

This section addresses the effect of the 17x17 NGF design on the LOCA-related analyses, up to the current lead rod average burnup limit of 62,000 MWD/MTU. Referring to Section 2.3, the following new features associated with the 17x17 NGF design need to be evaluated:

- Mid-grids with an I-spring rod support,
- Five Intermediate Flow Mixing (IFM) grids,
- New mixing vane design for mid-grids and IFM grids,
- Debris Filter Bottom Nozzle (DFBN) with reduced pressure drop features,
- Oxide coating for mitigation of debris related failures (already used on some plants), and
- Optimized ZIRLO™ fuel rod cladding.

These design features primarily affect the following aspects of the LOCA-related analyses:

1) Fuel assembly loss coefficient/pressure drop,
2) Spacer and IFM grid geometry (blocked area ratio, open area fraction, inner strap thickness and inner strap height) and number of grids,
3) Core flow distribution during transition cycles.

The 17x17 NGF design will utilize Optimized ZIRLO™ cladding, which will be evaluated as described in Reference 20 and is not considered further here. The variations in fuel rod design identified in Section 1.1 (i.e., fuel rod plenum spring clip, stack height adjustments, and burnable absorber variations) would be evaluated on a plant-specific basis and are not discussed further here.

The following LOCA-related analyses are addressed in this section:

- Section 5.2.2 – Large Break LOCA
- Section 5.2.3 – Small Break LOCA
- Section 5.2.4 – Post-LOCA Long Term Core Cooling
- Section 5.2.5 – Transition Core Evaluation

5.2.2 Large Break LOCA

For plants transitioning to the 17x17 NGF design, a Large Break LOCA analysis will be performed using one of the Westinghouse realistic methods (e.g., References 49, 50 or 51). All of the Westinghouse realistic methods utilize the NRC-approved WCOBRA/TRAC computer code, which has explicit models for fuel assembly hydraulic resistance and spacer grid heat transfer. As such, the changes in fuel assembly loss coefficient/pressure drop, grid geometry and number of grids for the 17x17 NGF design can be handled through appropriate specification of the WCOBRA/TRAC input.

Due to the addition of two IFM grids, the distance between grids in the corresponding spans is reduced relative to the previous designs. Since the core axial noding in WCOBRA/TRAC is dependent on the location of the spacer grids, the core axial noding for analyzing a full core of the 17x17 NGF design will be specified to accommodate the differences in grid spans due to the two additional IFMs. As with current designs, the continuity cell placement for a full core of the 17x17 NGF design will be determined using the basic approach described in Section 20-1-2 of Reference 50.

A sample WCOBRA/TRAC core noding diagram for a 17x17 RFA assembly and a 17x17 NGF assembly is provided in Figure 5.2-1. (Note that all values are rounded to the nearest 0.1 inch.) As can be seen from Figure 5.2-1, the differences between the sample 17x17 RFA and 17x17 NGF core noding occur in the [

]^{a,c}.

Since the 17x17 NGF fuel assembly has a higher pressure drop, a transition core evaluation will also be performed to assess the effect of flow redistribution on the 17x17 NGF assemblies during the transition cycles. To facilitate a direct quantification of the transition core effects, the core axial noding for these supplemental calculations will be modified as required for consistency between the resident and the 17x17 NGF fuel types. The results of the evaluation will determine the transition core effect that will be applied to the full-core 17x17 NGF case to establish the overall results for the 17x17 NGF design during the transition cycles.

5.2.3 Small Break LOCA

Small break LOCA calculations for PWR licensing applications are performed using the 1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP (NOTRUMP-EM)[52][53]. Small Break LOCA transients are characterized by a gradual top-down draining of the reactor coolant system, with low flow rates in the core relative to those occurring at steady-state or for Large Break LOCA transients. The hydraulic losses in the core due to frictional drag, form loss, and acceleration are small, and reasonable variations in flow resistance are expected to have a negligible effect on Small Break LOCA analysis results. These and other differences resulting from the 17x17 NGF design will be evaluated on a plant-specific basis.

5.2.4 Post-LOCA Long Term Core Cooling

The changes being introduced for the 17x17 NGF design are expected to have a minor effect on the Post-LOCA Long Term Core Cooling analyses and will be evaluated on a plant-specific basis.

5.2.5 Transition Core Evaluation

Section 5.2.2 outlines the transition core considerations for Large Break LOCA. The Small Break LOCA and Post-LOCA Long Term Core Cooling analyses are not sensitive to mixed-core effects, so no further consideration is required.

5.2.6 Conclusions

With respect to the LOCA-related analyses, the 17x17 NGF design features primarily affect the assembly loss coefficient/pressure drop, the spacer and IFM grid geometry, the number of grids, and the flow distribution during transition core cycles. Sections 5.2.2 to 5.2.4 outline the evaluation approach for the Large Break LOCA, Small Break LOCA, and Post-LOCA Long Term Core Cooling, respectively, and Section 5.2.5 summarizes the assessment of mixed-core effects.

5.3 LOCA Hydraulic Forces

5.3.1 LOCA Hydraulic Forces Overview

The following discussion focuses on calculations of LOCA hydraulic forces, and their effects on fuel and vessel internals qualification. Although other factors are considered in fuel qualification, such as seismic loading and component weight, the following discussion is primarily constrained to the generation and effects of hydraulic loads resulting from postulated pipe rupture.

The requirement to consider LOCA hydraulic forces in the design of fuel assemblies comes out of 10 CFR Part 50 Appendix A, General Design Criteria 4 (GDC-4) Environmental and Dynamic Effects Design Bases[54]:

> "Structures, systems, and components important to safety shall be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid systems piping rupture is extremely low under conditions consistent with the design basis for the piping."

The Standard Review Plan (SRP), NUREG-0800, Section 4.2, Revision 3 "Fuel System Design" Appendix A "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces"[55] provides specific guidance on appropriate methods for analysis of the fuel with respect to LOCA forces. The discussion of acceptance criteria for the fuel assembly response to externally applied forces denotes three primary areas of concern. Fuel rod fragmentation must not occur as a result of blowdown loads. 10 CFR 50.46 temperature and oxidation limits must not be exceeded. Control rod insertability for the combined seismic and LOCA loads must be demonstrated for the worst case LOCA that requires control rod insertion. More general guidance on postulated pipe rupture, such as break size, location, and break opening time is provided in SRP Section 3.6.2, Revision 2[56].

The LOCA hydraulic forces methodology Westinghouse uses in fuel assembly qualification was originally described in WCAP-9401-P-A[57] which was reviewed and approved by the NRC. These methods included postulated piping ruptures consistent with the locations specified in WCAP-8082-P-A[58], which was reviewed and approved by the NRC. The methods included the use of the MULTIFLEX computer code[59] along with its post-processors LATFORC and FORCE2 (Reference 59, Appendix A and B) which were reviewed and approved by the NRC for use in generating hydraulic forces resulting from PWR loss-of-coolant accidents.

The criteria and methods from References 55 through 59 considered limiting breaks that bounded the full spectrum of break sizes and locations in the main coolant loop piping. However, an issue was raised in 1975 regarding asymmetric cavity pressurization blowdown loads resulting from postulated pipe rupture and the validity of methods used to demonstrate compliance with GDC-4. This issue, which came to be referred to as Unresolved Safety Issue (USI) A-2, is described in Generic Letter 78-2[60]. As indicated in Reference 60, a number of potential solutions were proposed for investigation including in-service inspection, better analytical methods and more realistic assumptions, and probabilistic risk assessment. Many of these solution paths were pursued simultaneously, but ultimately the probabilistic assessment was approved, with the result that many postulated piping ruptures were excluded from the design basis for purposes of dynamic effects as allowed in GDC-4[54]. The first of these approvals was discussed in NRC Generic Letter 84-04[61]. Subsequent analyses performed for the same purpose after 1986, when Reference 54 was amended to allow for exclusion of piping from the GDC-4 design basis, have come to be referred to as "leak-before-break" (LBB). Currently, all Westinghouse designed US PWR primary coolant main loop piping has been excluded from consideration for dynamic effects associated with postulated pipe rupture under Reference 61 or subsequent LBB analyses. As a result, all current fuel qualification analyses are performed on the basis of postulated rupture of branch lines connected to the primary coolant loop.

Between 1981 and 1986, before it was clear that leak-before-break would receive NRC approval, Westinghouse developed and employed an improved version of MULTIFLEX, referred to as Version 3.0[62]. MULTIFLEX 3.0 differed from its predecessor in three respects, which made it more realistic. First, the series of one dimensional pipes representing the downcomer were replaced with a network of pipes to allow more realistic two-dimensional acoustic wave propagation in the downcomer. Second, the simple linear boundary conditions applied at the top and bottom of the barrel were replaced with more realistic non-linear boundary conditions to better reflect the real physical response to barrel motion. Third, the modal representation of the core barrel was modified to account for vessel motion relative to the barrel, rather than assuming a fixed rigid vessel. This combination of features provided a more realistic model of the hydraulic system response.

MULTIFLEX 3.0 had been submitted to the NRC for review in conjunction with a [ ] a, c fuel submittal[63]. The review was halted when the NRC began approving LBB in 1986. A number of plants had been analyzed using MULTIFLEX 3.0. A majority of fuel products that were analyzed with MULTIFLEX 3.0 were reanalyzed using MULTIFLEX (1.0), although [ ] a, c continued to rely on the MULTIFLEX 3.0 analysis on the basis of additional material submitted in Reference 64.

MULTIFLEX 3.0 was used in 1998 in analyses supporting new baffle-barrel-bolting patterns[65]. Although it was not specifically reviewed for generic application, it was accepted for use in the NRC approved baffle-bolting methodology. The primary success criteria for the baffle bolting program are the same as those documented in SRP Section 4.2[55] discussed above: i.e., no fuel fragmentation, 10 CFR 50.46 criteria continue to be met, and control rod insertability is maintained. These analyses were also based on LBB exclusion of the main coolant loop piping.

In 1999, MULTIFLEX 3.0 was used again in conjunction with control rod insertability analyses[66] performed for [ ] [a, c], which were reviewed and approved by the NRC[65], and included acceptance of the use of MULTIFLEX 3.0. As noted in Reference 67, although Reference 66 provided results for both main coolant loop piping breaks and branch line breaks, only the branch line breaks not covered by LBB are considered in the licensing basis. As a result of this analysis, [ ] [a,c] could credit control rod insertion for addressing boron dilution issues post-LOCA on the basis of branch line break LOCA loads.

The current methods for addressing the effects of postulated pipe rupture on fuel for compliance with GDC-4[54] as previously described are consistent with those presented in References 55 through 59 with the following exceptions: 1) the postulated pipe ruptures considered only include those branch line breaks not excluded under LBB licensing, and 2) MULTIFLEX 3.0 may be used for certain analyses in place of MULTIFLEX (1.0).

Analysis Assumptions

The key assumptions in a LOCA forces analysis include the primary reactor coolant system geometry (including fuel geometry), the full power operating conditions of temperature and pressure (including uncertainties), the postulated break sizes and locations (the largest branch lines not covered by LBB licensing exclusion) and the break opening time. (The break opening time is conservatively assumed to be 1 millisecond as stipulated in Reference 59, except in baffle bolt analyses as discussed in Reference 65). The hydraulic force calculations, being based on a network of one dimensional pipes, assume a full core of one particular fuel type. Transition core effects are addressed in the multiple fuel assembly models used in fuel qualification, as discussed in Reference 68.

17x17 NGF Fuel and LOCA Forces Codes and Methods

The significant features of the new fuel design in regard to LOCA forces are those that affect the assembly geometry, the assembly hydraulic resistances, and the assembly mechanical performance. The features that impact these include the two additional IFM grids, the modified 17x17 RFA mixing vanes, the reduced pressure drop DFBN, and the mid-grid I-spring rod support. While these features will require that the FORCE2 model account for the additional IFM grids, these features can be modeled using the existing codes and methods.

<u>Acceptance Criteria</u>

Although there are no specific requirements on the hydraulic forces themselves, the acceptance criteria for the 17x17 NGF fuel assemblies remain the same as those discussed in the introduction of this section: i.e., no fuel fragmentation is allowed under LOCA forces, the thermal and oxidation limits of 10 CFR 50.46 must continue to be met, and control rod insertability must be demonstrated for the largest break sizes postulated within the design basis to address operability for analyses that credit rod insertion.

5.3.2 LOCA Hydraulic Forces Reanalysis/Evaluation

The introduction of 17x17 NGF fuel will generally require reanalysis of LOCA hydraulic forces on a plant specific basis. The changes in geometry, hydraulic resistance, and mechanical performance each must be addressed, while accounting for differences in plant operating conditions, seismic spectra, and levels of LBB licensing.

As with current LOCA forces analyses, the largest cold leg break is postulated to produce the limiting horizontal loads on the vessel and fuel. Similarly, the largest hot leg break is postulated, primarily to produce the limiting vertical loads on the fuel.

5.3.3 LOCA Hydraulic Forces Conclusions

The 17x17 NGF fuel can be modeled with respect to LOCA forces using the existing methodology and computer codes. The changes in the assembly geometry, hydraulic performance, and structural strength and stiffness, along with differences in plant operating conditions, geometry, and seismic loads, will generally require reanalysis of LOCA forces on a plant specific basis to support the introduction of 17x17 NGF fuel.

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Figure 5.2-1
Sample WCOBRA/TRAC Continuity Cell Placement
for 17x17 RFA and 17x17 NGF Designs

a, c

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6.0 Reactor Vessel and Internals Evaluation

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor internals, fuel and control rod drive mechanisms. The reactor internals function to support and orient the reactor core fuel assemblies and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The reactor vessel internal components also function to direct coolant flow through the fuel assemblies (core), to provide adequate cooling flow to the various internals structures, and to support in-core instrumentation. They are designed to withstand forces due to structure deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads, earthquake accelerations, and Loss of Coolant Accident (LOCA) loads.

Reloading a reactor core with fuel other than that for which the plant was originally designed requires that the reactor vessel/fuel interface be thoroughly addressed to assure compatibility with the reactor vessel and internals and to assure that the structural integrity of the reactor vessel and internals are not adversely affected.

The areas potentially affected by a change in fuel are:

- Reactor Internals System Thermal-Hydraulic Performance
- Reactor Internals System Structural Response to Seismic and LOCA Conditions
- Rod Control Cluster Assembly (RCCA) Scram Performance
- Reactor Internal Components Structural Performance
- Reactor Internal Components Interface with the Fuel

6.1 Reactor Internals System Thermal-Hydraulic Performance

6.1.1 Introduction and Overview

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, i.e. vessel pressure drop, core bypass flow, and hydraulic lift forces. The pressure loss data is necessary input to the safety analysis and to the Nuclear Steam Supply System (NSSS) performance calculations. The hydraulic forces are critical in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold down spring, and the stresses in the reactor vessel closure studs.

6.1.2 Model and Methodology

The thermal-hydraulic analysis models the reactor vessel and internals system in a Westinghouse pressurized water reactors (PWR). The thermal-hydraulic analysis computes the following; the reactor vessel pressure losses for various flow rates, associated core bypass flows, baffle-barrel region flow rates, baffle joint momentum flux and baffle jetting margins of safety, baffle plate pressure relief hole velocities, hydraulic uplift forces, and hydraulic and geometrical data.

The reactor vessel pressure losses are calculated by classical analytical fluid mechanics. The thermal-hydraulic analysis solves the following continuity and momentum equations for a flow system that represented the entire reactor vessel and internals system.

Continuity

$$W = \rho VA = constant$$

where: W is the mass flow rate
ρ is the fluid density
V is the fluid velocity of the coolant
A is the area of the flow

Mechanical Energy

$$P_j = P_i + \sum_{i}^{j} (K + f\,l\,/\,D) \frac{\rho V^2}{2 g_C}$$

where: P is the pressure at i or j
K is the loss coefficient
f is the friction factor of the reactor internals
l is the characteristic length
D is the hydraulic diameter
ρ is the fluid density
V is the fluid velocity
g_c is the gravitational constant

Typically, in a pure analytical hydraulic analysis, the fluid properties appearing in the equations are known to a high degree of confidence. Therefore, if the pressure loss determination fails, it is generally due to the inability to analytically predict hydraulic loss coefficients of complex geometry. In order to eliminate this potential problem, these coefficients were experimentally determined from tests on $1/7^{th}$ scale pressurized water reactor models. Thus the pressure drops predicted in the thermal-hydraulic analysis are the results of standard hydraulic methods utilizing coefficients with a sound experimental basis.

The thermal-hydraulic analysis utilizes the fuel assembly design loss coefficients and geometric data as inputs. The fuel assembly data is used to determine the pressure drop across the core, which is essential in determining the reactor vessel pressure losses, core bypass flow, baffle-barrel region flow rates, baffle joint momentum flux and baffle jetting margins of safety, and hydraulic uplift forces.

The methodology used in the thermal-hydraulic analysis remains valid in the analysis of a Westinghouse pressurized water reactor implementing the 17x17 NGF fuel design. With the fuel change a plant-specific thermal-hydraulic analysis would be performed to address the impact of the fuel change. This plant-specific thermal-hydraulic analysis is performed to assure compatibility with the reactor vessel and internals and to assure that the structural integrity of the reactor vessel and internals are not adversely affected.

Preliminary thermal-hydraulic evaluations with 17x17 NGF have already been performed for several plants. The effect of the 17x17 NGF fuel design is a small increase in the pressure drop across the core. This small core pressure drop impacts the core bypass flow, baffle joint momentum flux and the hydraulic lift forces. The preliminary evaluations indicate that the 17x17 NGF will not cause an impact to the thermal-hydraulic analysis results such that the reactor internals design criteria are exceeded.

6.1.3 Conclusions

The methodology used in the thermal-hydraulic analysis of the reactor internals has been used on Westinghouse pressurized water reactors implementing changes in fuel. The thermal-hydraulic analysis methodology remains valid for Westinghouse pressurized water reactors implementing the 17x17 NGF fuel design. For any Westinghouse pressurized water reactor implementing 17x17 NGF, a plant-specific thermal-hydraulic analysis would have to be performed to address the impact of the 17x17 NGF fuel design.

6.2 Reactor Internals System Structural Response to Seismic and LOCA Conditions

6.2.1 Introduction and Overview

Changes in fuel assembly properties generally impact the performance of the reactor pressure vessel and its internals under all modes of operation. It is, therefore, important that with a change of fuel, the mechanical response of the reactor pressure vessel and its internals be evaluated. This is done to assure compatibility of the fuel with the reactor pressure vessel and its internals and to assure the structural integrity of the RPV system is not adversely affected. The mechanical system evaluations consist of response due to LOCA and response due to seismic excitations.

The mathematical system model of the RPV, internals and fuel is a three dimensional nonlinear finite element model. The only difference between the seismic and LOCA models is that, in the seismic model, fluid-solid interactions are represented by hydrodynamic mass matrices in the down-comer region. On the other hand, in the LOCA analysis the fluid-solid interactions are accounted through the hydraulic forcing functions with the use of the MULTIFLEX computer code (a fluid-solid interaction code).

The RPV system finite element model consists of three concentric structural sub-models connected by non-linear impact elements and linear stiffness matrices. The first sub-model represents the reactor vessel shell and associated components. The reactor vessel is restrained by reactor vessel supports situated beneath each nozzle and by the attached primary coolant piping. The second sub-model represents the reactor core barrel, baffle-former assembly, lower support plate, tie plates, secondary core support structure, etc. These sub-models are physically located inside the first, and are connected by stiffness matrices at the internals support ledges. Core barrel to reactor vessel shell impact is represented by non-linear elements at the core barrel flange, core barrel outlet nozzles, and the lower radial restraints. The third and innermost sub-model represents the upper support plate assembly consisting of guide tubes and upper support columns, upper and lower core plates, and the fuel. The third sub-model is connected to the first and second by stiffness matrices and non-linear elements.

The RPV system model utilizes the fuel assembly dynamic characteristics. The fuel assembly dynamic characteristics are provided through a simplified beam model. This fuel assembly simplified beam model is incorporated into the RPV system model to represent the dynamic characteristics of the fuel assembly and is used to determine the impact on the reactor internals and vessel.

A general-purpose finite element code is used to determine the response of the reactor vessel and its internals. In the finite element approach, the structure is divided into a finite number of discrete members or elements. The inertia and the stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Then, employing appropriate transformations, the element global matrices and arrays are assembled into global structural matrices and arrays, and used for dynamic solution of the differential equations of motion for the structure:

$$[M]\{\ddot{U}\} + [D]\{\dot{U}\} + [K]\{U\} = \{F\} \qquad (1)$$

where:

$[M]$	$\equiv$	the global inertia matrix
$[D]$	$\equiv$	the global damping matrix
$[K]$	$\equiv$	global stiffness matrix
$\{\ddot{U}\}$	$\equiv$	acceleration array
$\{\dot{U}\}$	$\equiv$	velocity array
$\{U\}$	$\equiv$	displacement array
$\{F\}$	$\equiv$	force array, including impact, thrust forces, hydraulic forces, constraints, weight and the like

Initial computer runs such as dead weight analysis and the vibrational (modal) analysis are made to set initial vertical gaps and to calculate eigenvalues and eigenvectors. The first time step of the time-history run performs the static solution due to operating hydraulic forces and gravity. After the initial time step, the dynamic solution of Equation (1) is calculated. Nodal displacements and impact forces are calculated and are used in the structural assessment of the fuel assembly and the reactor internals. The fuel assembly grid impact loads from the seismic core plate motions are combined with the grid impact loads from the LOCA core plate motions to evaluate the structural integrity of the fuel assembly. Also, an assessment of the impact forces generated at various interface locations in the RPV system model are made to ensure that the loads are within the acceptable limits.

6.2.2 Conclusion

The finite element modeling used in the LOCA and seismic analysis of the reactor vessel, reactor internals, and fuel is based on the NRC-approved approach documented in Reference 57. The approach of Reference 57 remains valid in the analysis of a Westinghouse pressurized water reactor implementing the 17x17 NGF fuel design. With the fuel change a plant-specific RPV system LOCA and seismic analysis would have to be performed to address the impact of the fuel change. This plant-specific RPV system LOCA and seismic analysis would be performed to assure compatibility with the reactor vessel and internals and to assure that the structural integrity of the reactor vessel and internals are not adversely affected by the change in fuel.

6.3 Rod Control Cluster Assembly (RCCA) Scram Performance

6.3.1 Introduction and Overview

The Rod Control Cluster Assemblies (RCCAs) represent one of the most critical interfaces between the fuel and the reactor internal components. Because of this critical interface it is necessary to ensure that the fuel does not adversely impact the operation of the control rods, either during accident conditions or normal operation.

6.3.2 Model and Methodology

During full power plant operation, all RCCAs and corresponding drive rod assemblies are held at a fully withdrawn position by respective Control Rod Drive Mechanisms (CRDMs). If any operation or accident event necessitates an immediate core power shutdown, all CRDMs will release the RCCAs, allowing them to drop from their fully withdrawn to their fully inserted positions. This gravity induced drop occurs only because the total RCCA/drive rod assembly weight is greater than the mechanical and hydraulic resistance forces opposing downward RCCA motion.

The excess of RCCA/drive rod assembly weight over resistive forces must be large enough that the drop, or SCRAM times are less than safety criteria specification for specific operating and external (e.g. earthquake) conditions.

In view of the expenditure involved in testing, a better economic position is realized by calculating actual SCRAM times for feasible driveline designs or design alterations, combined with various combinations of plant operating and external conditions, before full scale implementation of the designs. In this way, calculated SCRAM times can be compared against criteria to determine whether or not new designs or design alterations would be acceptable for implementation.

The fuel assembly design loss coefficients and geometric data are used as inputs to the RCCA scram analysis. The fuel assembly data is used in determining the fuel assembly drag force. The fuel assembly drag force along with the other mechanical and hydraulic resistance forces are used to determine the elapsed time for complete, gravity induced, RCCA insertion into the core.

The methodology used in determining RCCA scram time has been used on Westinghouse pressurized water reactors since its development in the 1970s. The RCCA scram time methodology remains valid for use in the analysis of a Westinghouse pressurized water reactors implementing the 17x17 NGF fuel design. With the fuel change, a plant-specific RCCA scram analysis would be performed to address the impact of the fuel change. This plant-specific RCCA scram analysis is performed to ensure that the fuel does not adversely impact the operation of the control rods, either during accident conditions or normal operation, and that the RCCA drop times are less than safety criteria specification.

Preliminary 17x17 NGF RCCA scram time evaluations have been performed for several plants. The results of these evaluations indicate that the RCCA scram times with 17x17 NGF increase slightly. This is primarily due to the increased fuel assembly pressure drop. Plant specific RCCA scram time analysis will be done to confirm that the RCCA scram time limit is not violated.

6.3.3 Conclusion

For any Westinghouse pressurized water reactor implementing 17x17 NGF, a plant-specific RCCA scram time analysis would have to be performed to address the impact of the fuel change.

6.4 Reactor Internal Components Structural Performance

6.4.1 Introduction and Overview

With a change in fuel design there is a possibility of increased gamma heating on the reactor internal components surrounding the core. Gamma heating has an impact on the structural integrity of the reactor internal components surrounding the core (i.e., upper and lower core plate and the components of the baffle-barrel region).

Structural evaluations are required to demonstrate that the structural integrity of the reactor internal components is not adversely affected by the change in fuel design. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth which must be accounted for in the design and analysis of the reactor internal components.

6.4.2 Model and Methodology

To evaluate the reactor internal components affected by increased gamma heating, a finite element model of the component is typically utilized. Using the finite element model, the effects of the increased thermal loading on the secondary stresses are determined. Moreover, a fatigue evaluation on cyclic stresses is typically performed as well. The model and methodology used in analyzing increased gamma heating on the reactor internal components is unaffected by the 17x17 NGF design.

6.4.3 Conclusion

For any Westinghouse pressurized water reactor implementing 17x17 NGF, a plant-specific reactor internal component thermal-structural analysis would have to be performed to address any increased gamma heating.

6.5 Reactor Internal Components Interface with the Fuel

6.5.1 Introduction and Overview

Reloading a reactor core with fuel other than that for which the plant was originally designed requires that the reactor internals/fuel interface be thoroughly addressed to assure compatibility with the reactor vessel and internals. This would require a review of all the interfaces between the fuel and reactor internal components. Change to the internals/fuel interfaces could result in changes to the loads applied to the reactor internal components. Any change to the applied loads could impact the structural integrity of the reactor internal component.

6.5.2 Model and Methodology

Typically, fuel design changes do not involve changes in the interfaces between the fuel and the reactor internal components. But when such a change does occur, the impact to the interfacing reactor internal component would have to be determined and evaluated. This would require a review of all the interfaces between the fuel and reactor internal components. The methodology used in evaluating such an impact would not be affected by the 17x17 NGF design.

6.5.3 Conclusion

For any Westinghouse pressurized water reactor implementing 17x17 NGF, a plant-specific review of the reactor internals/fuel interface should be performed to assure compatibility with the reactor vessel and internals.

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7.0 Radiological Assessment

The design changes that characterize the 17x17 NGF fuel assembly design consist of changes such as:

- Westinghouse Integral Top Nozzle (WIN) with removable top nozzle (RTN) feature
- Mid-grids with an I-spring rod support
- Five Intermediate Flow Mixing (IFM) grids
- New mixing vane design for mid-grids and IFM grids
- Debris Filter Bottom Nozzle (DFBN) with reduced pressure drop features
- Protective Grid
- Oxide coating for mitigation of debris related failures (already used on some plants)

The design changes for 17x17 NGF do not include changes to the fuel pellets themselves, nor to the fuel enrichment or fuel burnup limits. Subsequently, there is no direct impact on the core source term and no projected impact on the potential releases of activity from an operating plant during either normal operation or under accident conditions.

Thus, from a radiological assessment perspective, the 17x17 NGF fuel is essentially the same as the VANTAGE + fuel design which was assessed and approved in WCAP-12610-P-A[1] with the exception that the evaluation provided in WCAP-12610-P-A included consideration of extended fuel burnup up to [ ] a, c MWD/MTU and this application does not consider fuel burnup beyond the current licensed burnup limit of 62,000 MWD/MTU.

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